VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

October 24, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. Darrell G. Eisenhut,
Acting Director
Division Of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 760
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Docket Nos: 50-280
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50-338
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NPF-4

Dear Mr. Denton:

This is in response to your letter of September 13, 1979, regarding followup actions resulting from the NRC Staff review of the Three Mile Island Incident.

Our response for Surry Power Station Units 1 and 2 is included in Attachment A. Our response for North Anna Power Station Unit 1 is included in Attachment B.

Very truly yours,

C. M. Stallings

C. M. Stallings
Vice President-Power Supply
and Production Operations

cc: James P. O'Reilly

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Attachment A

Surry Power Station Units 1 and 2

Response to NRC Short Term
Requirements Resulting from the
Three Mile Island Incident

Section A1	Responses to items (c), (d) and (e) of September 13, 1979 letter
Section A2	Responses to requirements of NUREG 0578
Section A3	Response to Enclosure 7 of September 13, 1979 letter, on emergency preparedness

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RETURN TO REACTOR DOCKET
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Section A1

Response to items (c), (d) and (e) of NRC letter of September 13, 1979, for Surry Power Station.

NRC Position

(c) The ACRS comments on the shift technical advisor have resulted in our reassessment of the possible means of achieving the two functions which the Task Force intended to provide by this requirement. The two functions are accident assessment and operating experience assessment by people onsite with engineering competence and certain other characteristics. We have concluded that the shift technical advisor concept is the preferable short-term method of supplying these functions. We have also concluded that some flexibility in implementation may yield the desired results if there is management innovation by individual licensees. We have prepared a statement of functional characteristics for the shift technical advisor that will be used by the staff in the review of any alternatives proposed by licensees. A copy is provided as Enclosure 2.

Response

Beginning January 1, 1980 we will increase our manning to provide, on each shift, a shift technical advisor. Specific information regarding the shift technical advisors is included in our response to NUREG 0578 item 2.2.1.b.

NRC Position

(d) Three additional instrumentation requirements for short-term action were developed during the ACRS review of NUREG-0578. These items relate to containment pressure, containment water level and containment hydrogen monitors designed to follow the course of an accident. Descriptions of these items are provided in Enclosure 3.

(Enclosure 3)

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain containment conditions during the course of an accident, the following requirements shall be implemented:

(1) A continuous indication of containment pressure shall be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

(2) A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

(3) A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWR's and cover the range from the bottom to the top of the containment sump. Also for PWR's, a wide range instrument shall be provided and cover the range from the bottom of the containment to the elevation equivalent to a 500,000 gallon capacity. For BWR's, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

The containment pressure, hydrogen concentration and wide range containment water level measurements shall meet the design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy and testability. The narrow range containment water level measurement instrumentation shall be qualified to meet the requirements of Regulatory Guide 1.89 and shall be capable of being periodically tested.

Response

(1) The present design has four channels (PI-LM-100A-D) of protection grade containment pressure transmitters with indication in the control room. Their power supply is from Vital Buses I, II, III, IV, respectively. The range of indication is 0-65 PSIA. The lower range requirement is met, however, the upper range of indication must be increased to approx. 180 PSIA. (59.7 PSIA x 3 = 180). This will be evaluated and the appropriate design completed by July 1980. The installation will be completed during the next scheduled refueling outage for each unit.

(2) Our present design has two H_2 analyzers (H_2-A-GW-103,203) (located in the Auxiliary Bldg.), capable of being lined up to either containment. Their range of indication is 0 to 10% H_2. Their power supply is from Vital Buses VB1-II & VB2-II, respectively. Their operating pressure range is 9-18 PSIA and 60-150^{o}F temperature. These units must be manually lined up for operation following an accident. Either unit can monitor either containment. They meet the design and qualification provisions of Reg. Guide 1.97 including qualification, redundancy & testibility. This system will require modifications to provide remote isolation valves and control room readout. This will be accomplished by January 1, 1981.

(3) The present design has two containment water level channels (LI-RS-151A,B) which cover the range from the bottom of the containment to 40" span. We also have a containment sump monitor (LI-DA-100) with a range from the bottom to the top of the sump. The power supply are as follows: LI-RS-151A - Vital Bus. CH II; LI-RS-151B - Vital Bus CH III; LI-DA-100 - MB6-TB3 - (ISVB1 - MB6) ("J" Emergency Bus). All three channels readout in the control room. The height of water if 500,000 gals. were put into the containment would be > 40". Wide range transmitters will be installed by January 1, 1981.

NRC POSITION

(e) Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to insure the low probability of inadvertant actuation.

RESPONSE

A design change has been initiated and is estimated to require approximately 18 months to complete. This design will be expedited and the installation accomplished at the first scheduled outage of sufficient duration after material receipt.

Section A2

Response to recommendations of NUREG 0578 for Surry Power Station.

TITLE: Pressurizer Heater Power Supply (Section 2.1.1.3.1)

NRC POSITION

1. The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

RESPONSE

1. Two of the five pressurizer heater groups and their associated controls are fed from separate redundant safety related power sources which are capable of being fed from either the offsite or emergency power source. The NSSS vendor indicates that 125 kw is needed to provide natural circulation. The two backup heater groups are rated at 250 and 200 kw.

2. Procedures will be revised to instruct the operator in the use of pressurizer heaters in establishing and maintaining natural circulation. These revisions will be made in conjunction with overall review and revision of emergency procedures being conducted in response to NUREG 0578 item 2.1.9.

3. The backup groups are normally connected to the emergency buses in the existing heater control scheme. This existing control scheme ensures the timely initiation and maintenance of natural circulation conditions.

4. A review of the qualification of the pressurizer heater motive and control power equipment will be completed by March 31, 1980. If this review indicates the need for any modifications, we will provide details and a schedule for modification following completion of the review.

TITLE: Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators (Section 2.1.1.3.2)

NRC POSITION

1. Motive and control components of the power-operated relief valves (PORV's) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORV's and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. These buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

RESPONSE

Position 1 Control and indication circuits for the PORV's are powered from redundant safety grade emergency buses. Motive power for the power operated relief valves (PORV's) 2455C and 2456 is normally provided by containment instrument air. High pressure air supply tanks are provided as a redundant source of motive power for the PORV's as a part of the overpressurization system employed during solid water operation. Motive power for the PORV's has been upgraded by the addition of the high pressure air supply system. This will provide another source of motive power. The high pressure air supply system is seismically supported and is sized for 120 valve operations.

Positions 2,3, and 4
Motive power and control for the block valves is from redundant safety grade emergency buses.

Electrical motive and control power to the PORV's and associated block valves is qualified safety grade.

The pressurizer level indication instrument channels are powered from vital buses that are powered from redundant safety grade emergency buses.

A review of the qualification of the motive and control power equipment for the PORV's, block valves and the pressurizer level instrumentation will be completed by March 31, 1980. If this review indicates the need for any modifications, we will provide details and a schedule for modification following completion of the review.

TITLE: Relief and Safety Valve Testing (Section 2.1.2)

NRC POSITION

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test procedures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry piping and supports as well as the valves themselves.

RESPONSE

Vepco is participating in an Owners Group formed by utilities utilizing Westinghouse reactors. The Westinghouse Owners Group is working with other PWR owners and the Electric Power Research Institute (EPRI) to develop a program for qualification of relief and safety valves under expected operating conditions involving solid-water and two-phase flow conditions. The program description and schedule will be submitted by the required date of January 1, 1980.

Information developed from the test program will be utilized in a review of the pressurizer relief and safety valve design and piping configurations at Surry.

TITLE: Direct Indication of Valve Position (Section 2.1.3.a)

NRC POSITION

Reactor system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

RESPONSE

Pressurizer power operated relief valves 2455C and 2456 have direct indication derived from limit switches on each valve that indicate open-close position and are displayed in the control room. Each indication is powered from a vital source. In addition, an indirect indication is available by temperature indication of the P.O.R.V. discharge header. Motor operated block valves 2535 and 2536 in series with the P.O.R.V.'s have direct indication derived from limit switches on each valve that indicate open-close position and are displayed in the control room. The indication is powered from a vital source.

No direct indication of safety valve position exists. Indication of valve position through indirect means is available by temperature indication in the control room of the discharge pipe downstream of each valve.

We have evaluated two possible methods of monitoring the position of the safety valves. These methods are 1) direct indication of the safety valve position by mounting a limit switch on the safety valve, and 2) monitoring the flow in the discharge pipe with acoustic devices. We believe that the acoustical method is superior to the use of limit switches. We have requested proposals from vendors and should place an order for the necessary equipment prior to January 1980. We have started the preliminary design for the installation of the transducers, cabling, and monitoring cabinet.

Due to the delivery schedules for this equipment, installation of acoustic monitoring of the safety valves may require several months. In the interim, the following indications of safety valve positions are available.

1. Temperature indication is provided on the discharge piping of each safety valve.

2. Both temperature and level in the pressurizer relief tank are monitored.

3. With decreasing pressurizer pressure with indication that the PORV is closed, the operator would take action with emergency procedure "Loss of Coolant Accident".

TITLE: Instrumentation for Detection of Inadequate Core Cooling in PWR's and BWR's (Section 2.1.3.b.)

NRC POSITION

1. Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that the meter is not to be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

RESPONSE

1. Changes to emergency procedures have been made to emphasize the need to insure adequate coolant flow and to insure that reactor coolant temperature and pressure are maintained or immediately restored to achieve a margin to saturation of at least 50°F. All licensed reactor operators and current trainees have received special instruction in the TMI type incident with particular emphasis on the use of existing instrumentation to determine core conditions.

   We are participating in a Westinghouse Operating Plant Owners Group to more effectively deal with the generic issues resulting from the TMI incident. Included in the scope of work currently authorized by the group is a complete review and rewriting of the Westinghouse Generic Emergency Operating Instructions to incorporate lessons learned from the TMI incident. Specific concerns to be addressed in this review include the use of existing instrumentation to determine core conditions and the adequacy of core cooling. The scope and scheduling of this work is to be handled through the Bulletins and Orders Task Force. Improvements developed as part of the generic procedures review will be incorporated into the Surry emergency procedures. While the specific procedural applications of this owner's group effort are incomplete, the Westinghouse Owner's Group has developed an identification and categorization of those instruments which are essential for diagnosis of off normal conditions. The minimum set of instrumentation that is required for operator information in order to diagnose the type of plant event, take the necessary manual actions, and to monitor critical parameters is as follows.

Wide Range Reactor Coolant System Pressure
Wide Range RTD - hot legs
Wide Range RTD - cold legs
Pressurizer Level
Incore Thermocouples
RWST Level
Containment Sump Level
High Head Safety Injection Flow
Auxiliary Feedwater Flow
Condensate Storage Tank Level
Containment Pressure
Containment Radiation
Steamline Pressure
Steam Generator Narrow Range Level
Steam Generator Wide Range Level
Air Ejector Radiation
Steam Generator Blowdown Radiation
Boric Acid Tank Level
Control Room and Auxiliary Building Area
Radiation Monitors

Surry currently has all of this instrumentation.

We are developing a design for a reactor coolant saturation meter. We will make every effort to have this meter installed as soon as possible, with a target completion date of January 1, 1980. However, this date may be unattainable due to design and procurement uncertainties.

Mechanisms have been provided to allow the operator to immediately assess the primary coolant's margin to saturation conditions.

An operator can initiate a trend of system saturation temperature and one or more of the above listed temperatures. A saturation curve has been posted on the control board and provided in current procedures. This curve, combined with nearby indications of reactor coolant system temperatures and pressures, enables the operator to quickly determine the system's margin to saturation.

2. The identification of the need for any additional instrumentation will be made in conjunction with the ongoing analyses and procedural reviews. At this time, no definite additions or modifications have been determined to be necessary.

TITLE: Diverse Containment Isolation (Section 2.1.4)

NRC POSITION

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall describe the basis for selection of each essential system shall modify their containment isolation designs accordingly, and shall report the results of the reevaluation to the NRC.

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

RESPONSE

1. The Surry containment Phase 1 isolation system complies with the signal diversity requirements of SRP-6.2.4. All non-essential systems having automatic containment isolation valves and not required for an orderly reactor shutdown or to maintain containment atmospheric conditions, are closed on Phase 1 isolation which is initiated by a safety injection actuation signal. Safety injection is actuated by any one of the following input parameters: (1) high steam line flow with either low steam line pressure or low low T average, (2) high steam line differential pressure or (3) low pressurizer pressure, (4) high containment pressure or (5) manual actuation. These sources provide for the required diversity of parameters sensed which is in conformance with Section 6.2.4. of the Standard Review Plan.

2. Tables I through III list the essential and non-essential containment penetrations. The essential systems are divided into two categories (levels) which are based on their ability to mitigate the severity of various types of accidents. Level 1 of the essential systems are defined as Engineered Safety Features (ESF) and Containment Depressurization systems required to operate after a LOCA. These systems are listed in Table I.

   The essential Level 2 systems are defined as those required to maintain the operation of critical systems and functions such as the containment heat removal and, therefore, remain unisolated from the containment until a design basis LOCA is indicated (Phase 2 isolation) or when these systems are no longer required. These Level 2 systems are listed in Table II.

3. The non-essential systems listed in Table III are either isolated on Phase 1 actuation signal (SIS) or are closed during normal plant operation. Some non-essential systems listed in Table III may be utilized following a LOCA if conditions warrant.

   Our earlier review of containment isolation indicated the need for a design change to move the containment sump pump trip valve actuation from Phase II Containment Isolation to Phase I Containment Isolation. This design change has been completed.

4. Once Containment Phase 1 isolation has been initiated by a safety injection actuation signal, the automatic isolation valves can be opened only upon manual reset of the actuating signal and deliberate remote manual operation of the individual valve (refer to Section 6.2.4.3. of the FSAR).

TABLE I

ESSENTIAL SYSTEMS - LEVEL 1

System Description	Valve Position After SIS*
High Head Safety Injection to the Cold Leg	Open
High Head Safety Injection to the Hog Leg	Closed
Low Head Safety Injection to the Cold Leg	Open
Low Head Safety Injection to the Hot Leg	Closed
Low Head Safety Injection From the Sump	Closed
Containment Atmospheric Cleanup	Closed
Seal Water Injection to RC Pump	Open
Containment Spray Pump Discharge**	Closed**
Recirculation Spray Suction	Open
Recirculation Spray Discharge	Open
Service Water into the Recirculation Spray Heat Exchanger	Closed***

NOTES

* Isolation valves designated as closed receive a signal to close immediately after safety injection actuation and are opened by the operator or automatic controls at some period of time following a LOCA.

** Containment spray pump discharge valves are opened on high-high containment pressure. These valves can then be selectively closed after it is established that the system is no longer required.

*** Valves remain closed on SIS (Phase A) containment isolation. Valves open on a Containment Depressurization Actuation (CDA), containment isolation Phase B.

TABLE II

ESSENTIAL SYSTEMS - LEVEL 2

System Description	Mode of Containment Isolation
Component Cooling from RC Pump Thermal Barriers	Phase 2
Component Cooling from Containment Air Recirculation Cooling Coils	Phase 2
Component Cooling to RC Pump Motor	Phase 2
Component Cooling from RC Pump Motor	Phase 2
Main Steam Relief	Set Point Pressure
Auxiliary Feedwater	*
Component Cooling Water Return from RHR Heat Exchanger	Phase 2

* Closed system inside and check valve outside provide containment isolation.

TABLE III

NON-ESSENTIAL SYSTEMS

System Description	Mode of Containment Isolation
Charging-CVCS	Phase 1
Charging System Letdown	Phase 1
RC Pumps Seal Water Return	Phase 1
Containment Air Radiation Monitor Sample	Phase 1
Pressurizer Relief Tank Gas and Liquid Space Samples	Phase 1
Primary Coolant Hot Leg Sample	Phase 1
Primary Coolant Cold Leg Sample	Phase 1
Pressurizer Vapor Space Sample	Phase 1
Residual Heat Removal Sample	Phase 1
Containment Instrument Air Return	Phase 1
Safety Injection Accumulator Makeup	Adm. Controlled - Normally Closed
RHR Return to RWST	Adm. Controlled - Normally Closed
Steam Gen. Wet Layup	Adm. Controlled - Normally Closed
Primary Drain Transfer Discharge	Phase 1
Containment Sump Pump Discharge	Phase 1
Steam Gen. Blowdown	Phase 1
Service Air	Adm. Controlled - Normally Closed
Primary Grade Water	Phase 1
RC Loop Fill	Adm. Controlled - Normally Closed
Primary Vent Header	Phase 1

TABLE III (CONT'D.)

System Description	Mode of Containment Isolation
Nitrogen to Pressurizer Relief Tank	Phase 1
Primary Vent Pot Vent	Adm. Controlled - Normally Closed
Containment Leakage Monitoring	Phase 1
Condenser Air Ejector Vent	Phase 1
Containment Purge Air Ducts	Adm. Controlled - Normally Closed
Containment Air Ejector Suction	Locked - Closed
Pressurizer Dead Wt. Calibrator	Adm. Controlled - Normally Closed
Refueling Purifier Inlet and Outlet	Adm. Controlled - Normally Closed
Accumulator Tanks Test Line	Phase 1
Feedwater Chemical Addition	*
Main Steam (TRIP) - Shares Pen. wth M.S. Relief Lines	**
Feedwater	***
Auxiliary Feedwater	***

NOTES:

* Closed system inside and check valve to provide containment isolation.

** Isolated high steam line flow with low steam line pressure, or low low T average.

*** Closed system inside and check valves inside and outside provide containment isolation.

TITLE: Dedicated H_2 Control Penetration (Section 2.1.5.a)

NRC POSITION

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

RESPONSE

Surry has two installed H_2 recombiners in each containment. They are manually controlled from the Control Room Annex and placed into operation by procedure following a LOCA. In addition, provisions exist for the use of temporary skid-mounted recombiners. The temporary recombiner would take a suction on the containment through the same containment penetration which is used for the suction of the containment vacuum pumps. The recombiner discharges back to the containment through its own dedicated penetration. The suction penetrations are considered to be dedicated to the hydrogen recombiner during accident conditions since the containment vacuum system is not required for containment depressurization during accident conditions. The recombiner system is in no way connected to the purge system. The containment isolation for these penetrations meets the redundancy and single failure criteria. Deviations from General Design Criteria 54 and 56 are necessary to provide access to the automatic trip valves in order to ensure operability of the hydrogen recombiner and are documented in Section 6 of the FSAR. The two inch hydrogen recombiner lines tie into the two inch containment vacuum lines downstream of the containment isolation valves. To establish operation of the recombiner system, a temporary unit is brought in and connected to the post accident blowers located in the Auxiliary Building near the containment vacuum pumps. They must be locally controlled. This can be accomplished without excessive exposure to plant operating personnel. We have concluded that the existing arrangement satisfies the requirements of NUREG 0578.

TITLE: Inerting BWR Containments (Section 2.1.5.b)

NRC POSITION

It shall be required that the Vermont Yankee and Hatch 2 Mark I BWR containments be inerted in a manner similar to other operating BWR plants. Inerting shall also be required for near term OL licensing of Mark I and Mark II BWRs.

RESPONSE

This item is not applicable to Surry.

TITLE: Capability to Install Hydrogen Recombiner at Each Light Water Nuclear Power Plant (2.1.5.c)

NRC POSITION

1. All licensees of light water reactor plants shall have the capability to obtain and install recombiners in their plants within a few days following an accident if containment access is impaired and if such a system is needed for long-term post-accident combustible gas control.

2. The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

RESPONSE

Surry has two (2) Post Accident Hydrogen Recombiners rated at 50 scfm each that are located in each containment. The recombiners are separate and independent, and are manually controlled from the Control Room Annex. PT-31 provides operability testing on a periodic basis. The procedures and bases upon which the recombiners would be used have been issued and they place the recombiners into operation following Loss of Coolant incidents. As stated in response to item 2.1.5.a., provisions exist for the use of temporary skid-mounted recombiners.

TITLE: Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems) for PWR's and BWR's 2.1.6.a.

NRC POSITION

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate Leak Reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction

   Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

RESPONSE

A prerequisite to the establishment of the requested leakage program is the identification of those systems located outside containment which are likely to contain primary coolant immediately following accident. This will include portions of the accident mitigation systems located outside containment (Safety Injection System, Recirculation Spray System).

We are currently conducting a review of other systems to identify any additional systems which may be used in response to or recovery from an accident. A similar parallel review of systems use under post accident conditions is being conducted by Westinghouse for the Owners Group. It is expected that these reviews will be completed by January 1, 1980.

A leak reduction program for the accident mitigation systems will be developed and implemented by January 1, 1980. If the above mentioned reviews indicate the need for leakage reduction programs in other systems, these additional leakage reduction programs will be developed by March 1, 1980 and implemented as soon as operation permits.

Following is relevant information on the location and functions of the Residual Heat Removal System, Chemical and Volume Control System, Safety Injection System, and Recirculation Spray System. The Residual Heat Removal System (RHR) is located entirely within the containment. The RHR system does not perform any ESF functions. The Chemical and Volume Control System (CVCS) is leak tested as a portion of the Reactor Coolant System (RCS) and is isolated by a safety injection actuation. This leakage testing is governed by Technical Specification Table 4.1.2.A and is performed at least

once each day. Portions of other systems such as the Safety Injection System (SI), and the Recirculation Spray System (RS) are located outside the containment. The RS system is located within the Safeguards Building and the containment where the need for personnel access would be minimal. The outside recirculation spray system is normally aligned for operation and forms part of the containment boundary. Since the Surry units use subatmospheric containments, any significant leakage in the outside recirculation spray system would be obvious during operation due to its impact on establishing or maintaining a vacuum. The Low Head Safety Injection (LHSI) pumps are also located within the Safeguards Building where LHSI lines, when in the recirculation phase, are returned directly to the containment without traversing another building. The LHSI lines to the charging pumps (or High Head Injection Pumps) suction are directed through portions of the Auxiliary Building where the charging pumps are located.

TITLE: Plant Shielding Review (Section 2.1.6.b)

NRC POSITION

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4, each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls.

RESPONSE

A plant shielding review will be performed to evaluate the ability to operate essential systems required after a LOCA with significant core damage. This review will evaluate activity levels in areas of the plant which must be accessible to permit operation of essential systems and to ensure that safety grade equipment can perform its intended function in the resulting radiation field. Design changes, increased permanent or temporary shielding and/or post accident procedural controls will be implemented, where required, to assure the proper operation of vital systems.

As with the leakage reduction program, a prerequisite to the completion of the shielding review is the identification of those systems which are likely to contain highly radioactive materials following an accident. As explained in our response to item 2.1.b.a., internal and Owners Group studies are underway to determine which systems, in addition to mitigation systems should be included within the scope of the shielding and leakage reviews.

We are preceding with radiation level calculations for the accident mitigation systems (safety injection and recirculation spray). If the ongoing reviews indicate the need for a shielding review of additional systems, calculations for those system will follow. The identification of any shielding requirements or other corrective actions will be made following the completion of all radiation level calculations.

We expect the shielding review and identification of corrective actions to be completed by March 31, 1980. Implementation of corrective actions will begin by January 1, 1981.

TITLE: Auto Initiation of Auxiliary Feed (Section 2.1.7.a)

NRC POSITION

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term:

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses.

5. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

6. The a-c motor driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

RESPONSE

1. The current design of the Auxiliary Feedwater System provides for automatic initiation.

2. All initiation signals and circuits are designed to prevent a single failure from causing a loss of the Auxiliary Feedwater System.

3. The Auxiliary Feedwater System is initiated automatically by a safety injection signal, loss of offsite power, and on low-low steam generator level of any one steam generator. These actuation signals are testable and these signals are the system actuations on which the FSAR Chapter 14 accident analysis is based. The Auxiliary Feedwater System is also automatically initiated on loss of the main feedwater pumps in anticipation of low steam generator level. This anticipatory actuation is not testable during normal operation.

4. All initiating circuits which automatically start the Auxiliary Feedwater System, are powered from vital buses and are backed-up by the emergency power system.

5. The capability presently exists to manually initiate the Auxiliary Feedwater System from the control room. A single failure in the manual circuits will not result in the loss of system function.

6. The AC motor feed pumps in the Auxiliary Feedwater System are automatically initiated. The motor operated valves required to establish a flow path from the discharge of these pumps to the steam generators are left in the open position and also receive automatic signals. These valves are under strict administrative control and can be operated from the control room. The motor operated valves are powered from the emergency bus.

   The capability of cross-connecting auxiliary feedwater and supplying Auxiliary Feed from the unaffected unit has been installed at Surry. The valves receive automatic signals to open during certain steam rupture conditions. They are powered from the vital bus and are controlled manually from the control room. The same flow indications, individual steam generator isolation valve, and flow paths are utilized.

7. The automatic signals are designed in such a manner that their failure will not result in the loss of manual capability to start the Auxiliary Feedwater System. The automatic initiation circuits are presently safety-grade equipment and meet the long-term requirements.

TITLE: Auxiliary Feed Flow Indication (Section 2.1.7.b)

NRC POSITION

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

RESPONSE

1. Auxiliary Feedwater Flow indication to each steam generator which is displayed in the control room is safety grade equipment with the exception of power supply diversity.

2. Auxiliary Feedwater flow indication is powered from the emergency bus via the semi vital bus, which does not meet the diversity requirements of ASTB 10-1 of the standard review plan Section 10.4.9. To meet the diversity requirements of ASTB 10-1 the Auxiliary Feedwater flow indication power supples will be moved to an existing cabinet which meets the diversity requirements. This change will meet the position and will be implemented by January 1, 1980.

3. This change will result in safety-grade indication in the control room of auxiliary feedwater flow to each steam generator and thus satisfies all the requirements, including implementation category B, in NUREG-0578.

TITLE: Improved Post Accident Sampling Capability (Section 2.1.8.a)

NRC POSITION

A design and operational review of the reactor coolant and containment atmosphere sampling system shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremeties, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design change features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperature), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analysis shall be capable of being completed promptly; i.e., the boron sample analyses within an hour and the chloride sample analysis within a shift.

RESPONSE

A design and operational review of the reactor coolant and containment atmosphere sampling will be performed to determine the capability of personnel to promptly obtain samples under accident conditions without incurring a radiation exposure to any individual in excess of limits specified in 10 CFR 20. If this review indicates that samples cannot be promptly and safely obtained, we will provide a description of proposed modifications by January 1, 1980.

A design and operational review of the radiological spectrum analysis facilities will be performed to determine the capability of promptly quantifying core radioisotopes. If this review indicates that the required analysis cannot be performed promptly with the existing facilities, we will provide a description of proposed actions to meet the criteria by January 1, 1980.

These reviews represent preliminary reviews of general sampling capability. We have not addressed certain specific analysis stated in your position because we believe that additional clarification and discussion is needed to identify specific post accident time and sampling requirements. For example, determination of pH should be considered in preference to chlorides. We suggest that these requirements can best be addressed in a topical meeting with the Owners Group.

TITLE: Increased Range of Radiation Monitors (Section 2.1.8.b)

NRC POSITION

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

1. Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

   a. Noble gas effluent monitors with an upper range capacity of 10^5 uCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

   b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from a minimum of 10^{-7} u Ci/cc (Xe-133) to a maxmimum of 10^5 uCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors shall overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by absorption on charcoal or other media, followed by onsite laboratory analysis.

3. In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

RESPONSE

Noble gas effluent and radiation level monitors capable of providing the extended ranges recommended in Section 2.1.8.b do not appear to be commercially available at this time. Studies are being performed, however, to determine appropriate, conservative monitoring ranges based on plant specific parameters. These studies may show that commercially available equipment which approaches but does not reach the recommended range extremes can provide the necessary monitoring capability with an adequate margin of conservatism. Additional studies will be performed to determine how best to utilize existing station equipment for monitoring of radioiodines in gaseous effluents under accident conditions.

The findings of the above studies will be implemented to provide appropriate monitoring capability during the first refueling or extended outage following Jan. 1, 1981 or sooner if possible.

TITLE: Improved In-Plant Iodine Instrumentation (Section 2.1.8.c.)

NRC POSITION

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration throughout the plant under accident conditions.

RESPONSE

Improved capability for the assessment of in-plant airborne radioiodine concentrations under accident conditions will be provided by the following measures.

1. An adequate stock of "silver zeolite" sampling cartridges will be purchased and maintained at the station for emergency use. Existing station equipment will be used to perform gamma spectral analysis on collected samples to accurately assess iodine concentrations.

2. Procedures will be revised to instruct appropriate personnel in the proper precautions to be taken when sampling with charcoal cartridges. The procedure will address acceptable methods for removal of noble gases from charcoal cartridges, prior to performing gamma spectral analysis.

These measures will be implemented by January 1, 1980.

TITLE: Analysis of Design and Off-Normal Transients and Accidents (Section 2.1.9)

NRC POSITION

Analyses, procedures, and training addressing the following are required:

1. Small break loss-of-accident accidents;
2. Inadequate core cooling; and
3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in this appendix).

The analysis of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analysis shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator action is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors shall be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to date, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

RESPONSE

Analyses of small break loss-of-coolant accidents, symptoms of inadequate core cooling and required actions to restore core cooling, and analyses of transient and accident scenarios including operator actions not previously analyzed are being performed on a generic basis by the Westinghouse Owners Group. The small break analyses have been completed and were reported in WCAP-9600, which was submitted to the Bulletins and Orders Task Force by the Owners Group on June 29, 1979. Work required to address the other two areas, inadequate core cooling and other transient and accident scenarios, is being performed in conjunction with the Bulletins and Orders Task Force. Definitions of requirements and schedules for submittal of program results are being established with the Task Force. The results of these programs will determine if the need exists for any additional instrumentation or controls as required by Item 2.1.3.b.

In addition to the above program, the Owners Group is providing pre-test predictive analysis of the LOFT test program in accordance with the schedule established by the Bulletins and Orders Task Force.

The information from these analyses will be included in the plant emergency procedures. The initial Westinghouse review and revision of generic emergency procedures has been completed. The revised procedures were mailed to the NRC on October 16, 1979. Review of the revised procedures by Vepco training and operations personnel is currently in progress. Following acceptance of these procedures by the NRC, station specific procedures will be revised to incorporate the new generic guidelines. Operator training will be updated to include the bases and specifics of the revised procedures.

Any additional procedural changes determined to be necessary as a result of ongoing analyses will be implemented in a timely manner.

Our responses concerning containment pressure, water level, and hydrogen instrumentation and reactor vessel venting are included in Section A1.

TITLE: Shift Supervisors Responsibilities (Section 2.2.1.a)

NRC POSITION

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsiblity of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsiblity and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

   c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities and authority shall be clearly specified.

3. Training programs for shift supervisor shall emphasize and reinforce the responsiblity for safe operation and the management function the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsbility for assuring the safe operatin of the plant shall be delegated to other operations personnel not on duty in the control room.

RESPONSE

1.) A directive will be issued by the Vice President-Production, Operations and Maintenance which emphasizes the primary management responsiblity of the shift supervisor for safe operation of the plant under all conditions and clearly establishes his command duties. This directive will be issued prior to January 1, 1980 and at approximately yearly intervals thereafter.

2a,b,&c.) Existing administrative procedures delineate the responsibilities of station supervisory and operations personnel, including the authority of the shift supervisor. These procedures will be reviewed and revised by January 1, 1980 to include or emphasize the points cited in your position.

In addition, we will increase shift staffing by adding a SRO to allow the shift supervisor to maintain a broader oversight of plant safety and operating conditions. Shift staffing will then include two SRO's for one unit operation and three SRO's for two unit operation.

At least one of these senior reactor operators will be in the control room at all times. The shift supervisor will maintain an overview of plant conditions, make decisions regarding plant operations, and direct the actions of the control room operators.

3. Training programs for shift supervisors and SRO's will be improved by January 1, 1980 to provide greater emphasis on and reinforcement of the responsibility for safe operation and the management function the shift supervisor is to provide. Additional details on this training are included in our response to your position on Shift Technical Advisors.

4. The administrative duties of the shift supervisor will be reviewed by our Director of Nuclear Operations. This review will be completed by January 1, 1980. Additional control room manning, as discussed above, will allow the delegation of routine administrative duties to other control room personnel.

As explained in response to Item 2.2.1.b., the extra SRO is being added so that an SRO-Shift Technical Advisor will be on each shift. The additional SRO on each shift will begin on January 1, 1980.

TITLE: Shift Technical Advisor (Section 2.2.1.b)

NRC POSITION

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The shift technical advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The shift technical advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the shift technical advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

RESPONSE

As explained in Enclosure 3 of your September 13, 1979 letter, improvements are needed in the following two general areas:

1) Accident assessment capability
2) Review and utilization of operating experience.

Accident Assessment Capability

As explained in response to Item 2.2.1.a we intend to increase shift staffing to include an additional SRO. This additional staffing will begin on January 1, 1980. During 1980 a portion of the station SRO's will begin an advanced training program to qualify them as shift technical advisors. During 1980 at least one SRO who is participating in this training program will be on each shift and will be designated as the shift technical advisor (STA). In the event of an accident the shift technical advisor will withdraw from normal operational duties and will be dedicated to assessing plant conditions and advising the shift supervisor.

The shift technical advisors will complete a training program including the following subject:

1. Thermodynamics
2. Fluid Mechanics
3. Heat Transfer
4. Fuel Element Metallurgical Characteristics
5. Natural Circulation
6. Reactor Cooling Alternatives
7. Transients/Accidents* - performed on the Surry Simulator (including, but not limited to)
    a. FSAR accidents
    b. Low RCS pressure
    c. Low RCS flow
    d. Loss of coolant
    e. High RCS temperature
    f. High reactor power
    g. Uncontrolled cooldowns

* Will include multiple failures

It is estimated that this program will involve approximately 160 hours of classroom instruction and an equal time in independent study. Specific course details including content and scheduling will be finalized by January 1, 1980.

During 1980, those SRO's in training for shift technical advisor will complete the above training program so that beginning in January 1981 each shift will be manned with a fully trained shift technical advisor. If during 1980, due to manpower constraints an SRO/STA in training is not available, we reserve the option of substituting a degreed engineer with at least two years of nuclear power plant operating experience.

In the longer term we intend to upgrade SRO training to include the above training so that all SRO's will be qualified as shift technical advisors.

Review and Utilization of Operating Experience

Beginning not later than January 1, 1981, an individual with the appropriate technical knowledge and operating experience will be assigned full time to the review of internal and industry operating experiences. This person will initiate improvements to plant facilities or operational practices based on the knowledge and experience gained at our stations and throughout the industry. This individual will insure that the proper personnel including the STA's and training personnel receive the necessary reports and implement the lessons learned from them.

During 1980, due to manpower constraints, this function will be performed by the Operating Supervisor.

TITLE: Shift and Relief Turnover Procedures (Section 2.2.1.c.)

NRC POSITION

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of availabilty and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included on the checklist);

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic indepenent verification of system alignments).

RESPONSE

Presently, Administrative Procedures outline the requirements for shift turnover. This procedure will be revised by January 1, 1980 to incorporate signed off checklists to verify that systems important to safety are not in a degraded mode. The checklist will verify that primary plant parameters are in a normal band, and system alignments (ie. breaker controls and valve switches) are in accordance with the requirements of the mode of operation which the Unit is in. The checklist will require that the Minimum Equipment Status Record be reviewed with a verification of the time remaining until the Limiting Condition of Operation must be satisfied or the mode of operation changed.

Checklists will be implemented for the auxiliary operating stations which will require that they list maintenance activities in their area on safety related systems which could affect plant operation.

The checklists will require the signature of the oncoming and offgoing operator in each area and the oncoming shift supervisor. These checklists will be implemented by January 1, 1980.

The company quality assurance department conducts periodic audits and inspections to verify compliance with administrative controls including shift turnover procedures.

TITLE: Control Room Access (Section 2.2.2.a.)

NRC POSITION

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

RESPONSE

Existing administrative procedures establish the line of authority from the Station Manager through the Superintendent Operations, Operating Supervisor and the Shift Supervisor. The procedure is explicit in that it states that "the shift supervisor has the responsiblity of directing the actions of the station operators, to ensure safe and prudent operation of the facility". The Superintendent Operations, Operating Supervisor, Shift Supervisors and the Assistant Shift Supervisors are SRO's as mandated by Technical Specification 6.2.2.

The administrative procedures will be changed by January 1, 1980 to establish that the Shift Supervisor or Assistant Shift Supervisor has the authority and the responsibility to limit access to the control room during normal as well as emergency situations. The administrative procedure will establish clear lines of authority and responsiblity during emergency situations. Those persons in charge of the control room and with the authority to direct control room operations, shall be limited to Senior Reactor Operator licensed persons. The administrative procedure will clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

The administrative procedure will clearly limit access to those individuals responsible for direct operation of the station plus other technical advisors as needed by operations. Technical advisors, including non-Vepco personnel will be limited in number during any emergency or abnormal condition.

These administrative procedure changes will be completed by January 1, 1980.

TITLE: Onsite Technical Support Center (Section 2.2.2.b.)

NRC POSITION

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center.

Records that pertain to the as-built conditions and layout of structures, systems, and components shall be stored and filed at the site and accessible to the technical support center under emergency conditions. Examples of such records include system descriptions, general arrangement drawings, piping and instrument diagrams, piping system isometrics, electrical schematics, wire and cable lists and single line electrical diagrams. It is not the intent that all records described in ANSI N45.2.9-19-1974 be stored and filed at site and accessible to the technical support center under emergency conditions; however, as stated in the standard, storage systems shall provide for accurate retrieval of all pertinent information without undue delay.

RESPONSE

The Onsite Technical Support Center (OTSC) will be established in the Control Room Annex. The Annex is adjacent to the control room inside the protected security area, and is within the Control Room habitability envelope. The Annex has a complete controlled set of drawings, technical manuals, and other records which are properly stored and accessible. Communications are now available, including both NRC phone systems in the Annex as well as adequate commercial phones. The station PA system is also available in the Annex. By January 1, 1980, a remote typewriter accessing the Unit 1 and 2 computers will be placed in the OTSC. This typewriter can trend critical plant parameters for review by the technical support staff and consultants. We are currently investigating a more versatile permanent data link for the OTSC.

The Emergency Plan and Emergency Plan Implementing Procedures (EPIP's) will be revised to incorporate the role of the Technical Support Center. The Onsite Technical Support Center will be established as soon as possible, but no later than January 1, 1980. The Emergency Plan and EPIP's will be updated to reflect these changes by January 1, 1980.

TITLE: Onsite Operational Support Center (Section 2.2.2.c)

NRC POSITION

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

RESPONSE

Places of assembly at the onset of an emergency are designated by the Emergency Plan Implementing Procedures (EPIP's). The Emergency Director will announce that all persons are to report to their normal place of assembly for accountablity. This plan of action immediately establishes a situation whereby the shift supervisor or the Emergency Director can locate the needed assistance; i.e. H. P. Techs in the H.P. office, instrument techs in the instrument shop, etc.

The procedures will be changed whereby operators not required for control room operations will gather in the Emergency Switchgear room unless performing an operations function outside of the control room or otherwise instructed by the Shift Supervisor. Existing procedures now instruct other emergency teams such as the fire brigade and the first aid teams to assemble in the plant assembly room so as to be readily accessible to the Emergency Director. Other support groups such as Health Physics, Instrument Technicians, Chemistry Technicians, Engineers, and the maintenance personnel will assemble in their respective work areas.

The revised EPIP's will include explicit instructions that prohibit their entry into the control room unless requested by the shift supervisor. All of the above areas are served by adequate communications including commercial intraplant telephone and the station PA system. An intraplant telephone will be installed in the emergency switchgear room to allow better communications from the Control Room to the Operations Support Center. A station PA system is already available.

Procedures will be revised to reflect the above changes by January 1, 1980.

Section A3

Response to Enclosure 7 of September 13, 1979 letter, "Near Term Requirements for Improving Emergency Preparedness".

NRC POSITION

(1) Upgrade licensee emergency plans to satisfy Regulatory Guide 1.101, with special attention the development of uniform action level criteria based on plant parameters.

(2) Assure the implementation of the related recommendations of the Lessons Learned Task Force involving instrumentation to follow the course of an accident and relate the information provided by this instrumentation to the emergency plan action levels. This will include instrumentation for post-accident sampling, high range radioactivity monitors, and improved in-plant radioiodine instrumentation. The implementation of the Lessons Learned Task Force's recommendations on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.

(3) Determine that an emergency operations center for Federal, State and Local personnel has been established with suitable communications to the plant, and that upgrading of the facility in accordance with the Lessons Learned Task Force's recommendation for an in-plant technical support center is underway.

(4) Assure that improved licensee offsite monitoring capabilities (including additional thermoluminescent dosimeters or the equivalent) have been provided for all sites.

(5) Assess the relationship of State/local plans to the licensees' and Federal plans so as to assure the capability to take appropriate emergency actions. Assure that this capability will be extended to a distance of ten miles. This item will be performed in conjunction with the Office of State Programs and the Office of Inspection and Enforcement.

(6) Require test exercises of approved emergency plans (Federal, State, local and licensees), review plans for such exercises, and participate in a limited number of joint exercises. Tests of licensee plans will be required to be conducted as soon as practical for all facilities and before reactor startup for new licensees. Exercises of State plans will be performed in conjunction with the concurrence reviews of the Office of State Programs. As a preliminary planning basis, assume that joint test exercises involving Federal, State, local and licensees will be conducted at the rate of about ten per year, which would result in all sites being exercised once each five years. Revised planning guidance may result from the ongoing rulemaking.

RESPONSE

(1) The Emergency Plan that is currently in effect at Surry Power Station satisfies the requirements of Regulatory Gide 1.101.

(2) Our responses to the Lessons Learned Task Force recommendations regarding instrumentation to follow the course of an accident are included in Section A2 of this response. Existing emergency procedures use currently installed instrumentation in establishing emergency action levels. By January 1, 1981, following installation of additional instrumentation, emergency plans will be revised to include definitive criteria for emergency action levels using the new and existing instrumentation.

(3) The station training center has been designated as the emergency operations center for Federal, State and local personnel. This facility will be upgraded in conjunction with the Onsite Technical Support Center. We are currently evaluating locations for an alternate emergency center. An alternate center will be designated by January 1, 1980.

(4) Offsite monitoring capabilities will be reviewed and improved as required by January 1, 1980.

(5) We have reviewed state/local plans and are satisfied with their capability to take appropriate emergency action.

(6) Emergency exercises for site personnel are conducted yearly. Coordinated emergency exercises involving Federal, State, and local agencies will be conducted every 5 years.

Attachment B

North Anna Power Station Unit 1

Response to NRC Short Term
Requirements Resulting from the
Three Mile Island Incident

Section B1	Responses to items (c), (d) and (e) of September 13, 1979 letter
Section B2	Responses to requirements of NUREG 0578
Section B3	Response to Enclosure 7 of September 13, 1979 letter, on emergency preparedness

Section B1

Response to items (c), (d) and (e) of NRC letter of September 13, 1979, for North Anna Power Station.

NRC Position

(c) The ACRS comments on the shift technical advisor have resulted in our reassessment of the possible means of achieving the two functions which the Task Force intended to provide by this requirement. The two functions are accident assessment and operating experience assessment by people onsite with engineering competence and certain other characteristics. We have concluded that the shift technical advisor concept is the preferable short-term method of supplying these functions. We have also concluded that some flexibility in implementation may yield the desired results if there is management innovation by individual licensees. We have prepared a statement of functional characteristics for the shift technical advisor that will be used by the staff in the review of any alternatives proposed by licensees. A copy is provided as Enclosure 2.

Response

Beginning January 1, 1980 we will increase our manning to provide, on each shift, a shift technical advisor. Specific information regarding the shift technical advisors is included in our response to NUREG 0578 item 2.2.1.b.

NRC Position

(d) Three additional instrumentation requirements for short-term action were developed during the ACRS review of NUREG-0578. These items relate to containment pressure, containment water level and containment hydrogen monitors designed to follow the course of an accident. Descriptions of these items are provided in Enclosure 3.

(Enclosure 3)

Consistent with satisfying the requirements set forth in General Design Criterion 13 to provide the capability in the control room to ascertain containment conditions during the course of an accident, the following requirements shall be implemented:

(1) A continuous indication of containment pressure shall be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

(2) A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

(3) A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWR's and cover the range from the bottom to the top of the containment sump. Also for PWR's, a wide range instrument shall be provided and cover the range from the bottom of the containment to the elevation equivalent to a 500,000 gallon capacity. For BWR's, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

The containment pressure, hydrogen concentration and wide range containment water level measurements shall meet the design and qualification provisions of Regulatory Guide 1.97, including qualification, redundancy and testability. The narrow range containment water level measurement instrumentation shall be qualified to meet the requirements of Regulatory Guide 1.89 and shall be capable of being periodically tested.

Response

(1) The present design has four channels of protection grade containment pressure transmitters with indication in the control room. Their power supply is from Vital Buses I, II, III, IV, respectively. The range of indication is 0-60 PSIA. The lower range requirement is met. However, the upper limit of indication must be increased to approx. 180 PSIA. (59.7 PSIA x 3 = 180). This will be evaluated and the appropriate design completed by July 1980. The installation will be completed during the next scheduled refueling outage.

(2) Our present design has two H_2 analyzers capable of being lined up to either containment. Their range of indication is 0 to 10% H_2. These units must be manually lined up for operation following an accident. Either unit can monitor either containment. This system will require modifications to provide remote isolation valves and control room readout. This will be accomplished by January 1, 1981.

(3) The present design has two containment water level channels (LI-RS-151A,B) which cover the range from the bottom of the containment to 10 feet. Since the existing range encompasses over 500,000 gallons, no modifications are necessary.

NRC POSITION

(e) Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to insure the low probability of inadvertant actuation.

RESPONSE

A design change study has been initiated and is estimated to require approximately 18 months to complete. This design will be expedited and the installation accomplished at the first scheduled outage of sufficient duration after material receipt. Design and operational details will be submitted as soon as they are available.

Section B2

Response to recommendations of NUREG 0578 for North Anna Power Station.

TITLE: Pressurizer Heater Power Supply (Section 2.1.1.3.1)

NRC POSITION

1. The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

RESPONSE

1. Two of the five pressurizer heater groups are fed from separate redundant safety related 480 volt load centers. The NSSS vendor indicates that 125 kw is needed to provide natural circulation. The two backup heater groups are rated at 270 and 215 kw.

2. The pressurizer heaters may be connected to the emergency buses within the limitation of the diesel generator at any time following a loss of offsite power accident. If all loads that could be automatically connected to the emergency bus are connected, the heaters cannot be connected to the emergency bus after a loss of offsite power accident until a reduction in load has been accomplished. During natural circulation operation many of the emergency loads will not be connected. There is a kw meter on each of the emergency diesel generator control panels in the main control room, so that the operator can observe the diesel load and keep it within limits. Station operating procedures will be developed by January 1, 1980 for the load shedding sequences, and for instruction of the operator in the use of pressurizer heaters in establishing and maintaining natural circulation.

3. The NSSS vendor specifies that pressurizer heaters should be available within 1 hour in order to initiate and maintain natural circulation. Restoration of pressurizer heaters can be accomplished within a few minutes.

4. Motive and Control power interface equipment meets safety grade requirements except the supports for the tray sections that enter the bottom of the pressurizer. Although the cabling between the pressurizer heater distribution panels and the heaters themselves are not color-coded, the cabling is in separate raceways and meets the intent of the color-coded separation requirements in the FSAR. The tray section to the heaters will be upgraded to withstand seismic loads. Upgrading the tray sections to withstand seismic loadings will be accomplished by January 1, 1980.

TITLE: Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators (Section 2.1.1.3.2)

NRC POSITION

1. Motive and control components of the power-operated relief valves (PORV's) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORV's and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. These buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

RESPONSE

Position 1 Control and indication circuits for the PORV's are powered from redundant safety grade emergency buses. Motive power for the power operated relief valves (PORV's) 2455C and 2456 is currently provided only by containment instrument air. Nitrogen supply tanks are provided as a redundant source of motive power for the PORV's as a part of the overpressurization system employed during solid water operation. Motive power for the PORV's will be upgraded by the addition of a line and three (3) spring loaded check valves to the nitrogen supply system. This will provide another source of motive power. The nitrogen supply system is composed of seismically supported, stainless steel components and is sized for 120 valve operations. This modification will be completed on Unit 1 by January 1, 1980.

Position 2 Motive power and control for the block valves is from redundant safety grade emergency buses.

Position 3 Electrical motive and control power to the PORV's and associated block valves is qualified safety grade.

Position 4 The pressurizer level indication instrument channels are powered from vital buses that are powered from redundant safety grade emergency buses.

TITLE: Relief and Safety Valve Testing (Section 2.1.2)

NRC POSITION

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test procedures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry piping and supports as well as the valves themselves.

RESPONSE

Vepco is participating in an Owners Group formed by utilities utilizing Westinghouse reactors. The Westinghouse Owners Group is working with other PWR owners and the Electric Power Research Institute (EPRI) to develop a program for qualification of relief and safety valves under expected operating conditions involving solid-water and two-phase flow conditions. The program description and schedule will be submitted by the required date of January 1, 1980.

Information developed from the test program will be utilized in a review of the pressurizer relief and safety valve design and piping configurations at North Anna.

TITLE: Direct Indication of Valve Position (Section 2.1.3.a)

NRC POSITION

Reactor system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

RESPONSE

Pressurizer power operated relief valves 2455C and 2456 have direct indication derived from limit switches on each valve that indicate open-close position and are displayed in the control room. The indication is powered from a vital source. In addition, an indirect indication is available by temperature indication of the P.O.R.V. discharge header. Motor operated block valves 2535 and 2536 in series with the P.O.R.V.'s have direct indication derived from limit switches on each valve that indicate open-close position and are displayed in the control room. The indication is powered from a vital source.

No direct indication of safety valve position exists. Indication of valve position through indirect means is available by temperature indication in the control room of the discharge pipe downstream of each valve.

We have evaluated two possible methods of monitoring the position of the safety valves. These methods are 1) direct indication of the safety valve position by mounting a limit switch on the safety valve, and 2) monitoring the flow in the discharge pipe with acoustic devices. We believe that the acoustical method is superior to the use of limit switches. We have requested proposals from vendors and should place an order for the necessary equipment prior to January 1980. We have started the preliminary design for the installation of the transducers, cabling, and monitoring cabinet.

Due to the delivery schedules for this equipment, installation of acoustic monitoring of the safety valves may require several months. In the interim, the following indications of safety valve positions are available.

1. Temperature indication is provided on the discharge piping of each safety valve.

2. Both temperature and level in the pressurizer relief tank are monitored.

3. The P.O.R.V.'s are set to open before the safety valves, and the P.O.R.V.'s have direct indication. Should the safety valves actuate after the P.O.R.V's have actuated, and if blowdown, as evidenced by items 1 and 2, continues after indication of P.O.R.V. closure a malfunctioning safety valve would be recognized.

TITLE: Instrumentation for Detection of Inadequate Core Cooling in PWR's and BWR's (Section 2.1.3.b.)

NRC POSITION

1. Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that the meter is not to be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

RESPONSE

1. Changes to emergency procedures have been made to emphasize the need to insure adequate coolant flow and to insure that reactor coolant temperature and pressure are maintained or immediately restored to achieve a margin to saturation of at least 50°F. All licensed reactor operators and current trainees have received special instruction in the TMI type incident with particular emphasis on the use of existing instrumentation to determine core conditions.

   We are participating in a Westinghouse Owners Group to more effectively deal with the generic issues resulting from the TMI incident. Included in the scope of work currently authorized by the group is a complete review and rewriting of the Westinghouse Generic Emergency Operating Instructions to incorporate lessons learned from the TMI incident. Specific concerns to be addressed in this review include the use of existing instrumentation to determine core conditions and the adequacy of core cooling. The scope and scheduling of this work is to be handled through the Bulletins and Orders Task Force. Improvements developed as part of the generic procedures review will be incorporated into the North Anna emergency procedures. While the specific procedural applications of this owner's group effort are incomplete, Westinghouse has developed an identification and categorization of those instruments which are essential for diagnosis of off normal conditions. The minimum set of instrumentation that is required for operator information in order to diagnose the type of plant event, take the necessary manual actions, and to monitor critical parameters is as follows.

Wide Range Reactor Coolant System Pressure
Wide Range RTD - hot legs
Wide Range RTD - cold legs
Pressurizer Level
Incore Thermocouples
RWST Level
Containment Sump Level
High Head Safety Injection Flow
Auxiliary Feedwater Flow
Condensate Storage Tank Level
Containment Pressure
Containment Radiation
Steamline Pressure
Steam Generator Narrow Range Level
Steam Generator Wide Range Level
Air Ejector Radiation
Steam Generator Blowdown Radiation
Boric Acid Tank Level
Control Room and Auxiliary Building Area
    Radiation Monitors

North Anna currently has all of this instrumentation.

We are developing a design for a reactor coolant saturation meter. We will make every effort to have this meter installed as soon as possible, with a target completion date of January 1, 1980. However, this date may be unattainable due to design and procurement uncertainties.

Mechanisms have been provided to allow the operator to immediately assess the primary coolant's margin to saturation conditions.

An operator can initiate a trend of system saturation temperature and one or more of the above listed temperatures. A saturation curve has been posted on the control board and provided in current procedures. This curve, combined with nearby indications of reactor coolant system temperatures and pressures, enables the operator to quickly determine the system's margin to saturation.

2. The identification of the need for any additional instrumentation will be made in conjunction with the ongoing analyses and procedural reviews. At this time, no definite additions or modifications have been determined to be necessary.

TITLE: Diverse Containment Isolation (Section 2.1.4)

NRC POSITION

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall describe the basis for selection of each essential system shall modify their containment isolation designs accordingly, and shall report the results of the reevaluation to the NRC.

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

RESPONSE

1. The North Anna containment Phase A isolation system complies with the signal diversity requirements of SRP-6.2.4. All non-essential systems having automatic containment isolation valves and not required for an orderly reactor shutdown or to maintain containment atmospheric conditions, are closed on Phase A isolation which is initiated by a safety injection actuation signal. Safety injection is actuated by any one of the following input parameters: (1) high steam line flow with either low steam line pressure or low low T average, (2) high containment pressure, (3) high steam line differential pressure, (4) low pressurizer pressure, and (5) manual actuation. These four sources provide for the required diversity of parameters sensed which is in conformance with Section 6.2.4. of the Standard Review Plan.

2. Tables I through III list the essential and non-essential containment penetrations. The essential systems are divided into two categories (levels) which are based on their ability to mitigate the severity of various types of accidents. Level 1 of the essential systems are defined as Engineered Safety Features (ESF) and Containment Depressurization systems required to operate after a LOCA. These systems are listed in Table I.

   The essential Level 2 systems are defined as those required to maintain the operation of critical systems and functions such as the containment heat removal and, therefore, remain unisolated from the containment until a design basis LOCA is indicated (Phase B isolation) or when these systems are no longer required. These Level 2 systems are listed in Table II.

3. The non-essential systems listed in Table III are either isolated on Phase A actuation signal (SIS) or are closed during normal plant operation. Some non-essential systems listed in Table III may be utilized following a LOCA if conditions warrant.

4. Once Containment Phase A isolation has been initiated by a safety injection actuation signal, the automatic isolation valves can be opened only upon manual reset of the actuating signal and deliberate remote manual operation of the individual valve (refer to Section 6.2.4.3. of the FSAR).

TABLE I

ESSENTIAL SYSTEMS - LEVEL 1

System Description	Valve Position After SIS*
High Head Safety Injection to the Cold Leg	Open
High Head Safety Injection to the Hog Leg	Closed
Low Head Safety Injection to the Cold Leg	Open
Low Head Safety Injection to the Hot Leg	Closed
Low Head Safety Injection From the Sump	Closed
Containment Atmospheric Cleanup	Closed
Seal Water Injection to RC Pump	Open
Containment Spray Pump Discharge**	Closed**
Recirculation Spray Suction/Casing Cooling Discharge **	Open
Recirculation Spray Discharge	Open
Service Water into the Recirculation Spray Heat Exchanger	Closed***
Service Water Return From the Recirculation Spray Heat Exchanger	Closed***

NOTES

* Isolation valves designated as closed receive a signal to close immediately after safety injection actuation and are opened by the operator or automatic controls at some period of time following a LOCA.

** Containment spray pump discharge valves are opened on high-high containment pressure. These valves can then be selectively closed after it is established that the system is no longer required.

*** Valves remain closed on SIS (Phase A) containment isolation. Valves open on a Containment Depressurization Actuation (CDA), containment isolation Phase B.

TABLE II

ESSENTIAL SYSTEMS - LEVEL 2

System Description	Mode of Containment Isolation
Component Cooling from RC Pump Thermal Barriers	Phase B
Component Cooling from Containment Air Recirculation Cooling Coils	Phase B
Component Cooling to RC Pump Motor	Phase B
Component Cooling from RC Pump Motor	Phase B
Main Steam Relief	Set Point Pressure
Auxiliary Feedwater	*
Component Cooling Water Return RHR Heat Exchanger	Phase B
Containment Instrument Air	Phase B

* Auxiliary feedwater is isolated when the condensate storage tank has been depleted.

TABLE III

NON-ESSENTIAL SYSTEMS

System Description	Mode of Containment Isolation
Charging-CVCS	Phase A*
Charging System Letdown	Phase A*
RC Pumps Seal Water Return	Phase A*
Containment Air Radiation Monitor Sample	Phase A*
Pressurizer Relief Tank Gas and Liquid Space Samples	Phase A*
Primary Coolant Hot Leg Sample	Phase A*
Primary Coolant Cold Leg Sample	Phase A*
Pressurizer Vapor Space Sample	Phase A*
Residual Heat Removal Sample	Phase A*
Safety Injection Accumulator Makeup	Manual Locked-closed
RHR Return to RWST	Manual Lockedclosed
Steam Gen. Wet Layup	Manual Locked-closed
Primary Drain Transfer Discharge	Phase A*
Containment Sump Pump Discharge	Phase A*
Steam Gen. Blowdown	Phase A*
Service Air	Manual Locked-closed
Primary Grade Water	Phase A*
RC Loop Fill	Manual Locked-closed
Primary Vent Header	Phase A*
Nitrogen to Waste Gas Charcoal Filters	Phase A*

TABLE III (CONT'D.)

System Description	Mode of Containment Isolation
Nitrogen to Pressurizer Relief Tank	Phase A*
Primary Vent Pot Vent	Manual Locked-closed
Containment Leakage Monitoring	Phase A*
Steam Generator Blowdown Sample	Phase A*
Condenser Air Ejector Vent	Phase A*
Containment Purge Air Ducts	Locked-closed
Containment Air Ejector Suction	Locked - Closed
Pressurizer Dead Wt. Calibrator	Manual Locked-closed
Refueling Purifier Inlet and Outlet	Manual Locked-closed
Accumulator Tanks Test Line	Phase A*
Feedwater Chemical Addition	*
Main Steam (TRIP) - Shares Pen. with M.S. Relief Lines	**
Feedwater	***
Auxiliary Feedwater	***

NOTES:

* Valves may be opened by the control room operator at some time after a LOCA, if required.

** Closed system inside and check valve to provide containment isolation.

*** Isolated either by intermediate high-high containment pressure, high steam line flow with low steam line pressure, or low low T average.

TITLE: Dedicated H_2 Control Penetration (Section 2.1.5.a)

NRC POSITION

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

RESPONSE

The post-accident hydrogen recombiners take suction from the containment through the same containment penetration used for the suction of the containment vacumn pumps. The recombiner discharges back to the containment through its own dedicated penetration. The suction penetrations are considered to be dedicated to the hydrogen recombiner during accident conditions since the containment vacuum system is not required for containment depressurization during accident conditions. The recombiner system is in no way connected to the purge system. The containment isolation for these penetrations meets the redundancy and single failure criteria. Deviations from General Design Criteria 54 and 56 are necessary to provide access to the automatic trip valves in order to ensure operability of the hydrogen recombiner and are documented in Section 6 of the FSAR. The two inch hydrogen recombiner lines tie into the two inch containment vacuum lines downstream of the containment isolation valves. To establish operation of the recombiner system, a manual valve (not subject to single active failure by spurious motor-operator movement), which is accessible on the 274' elevation of the Auxiliary Building near the containment vacuumn pumps, must be closed to isolate the containment atmosphere from the containment vacuum system. This can be accomplished without excessive exposure to plant operating personnel. The existing arrangement, therefore, satisfies the intent of NUREG 0578.

TITLE: Inerting BWR Containments (Section 2.1.5.b)

NRC POSITION

It shall be required that the Vermont Yankee and Hatch 2 Mark I BWR containments be inerted in a manner similar to other operating BWR plants. Inerting shall also be required for near term OL licensing of Mark I and Mark II BWRs.

RESPONSE

This item is not applicable to North Anna.

TITLE: Capability to Install Hydrogen Recombiner at Each Light Water Nuclear Power Plant (2.1.5.c)

NRC POSITION

1. All licensees of light water reactor plants shall have the capability to obtain and install recombiners in their plants within a few days following an accident if containment access is impaired and if such a system is needed for long-term post-accident combustible gas control.

2. The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

RESPONSE

North Anna has two (2) Post Accident Hydrogen Recombiners rated at 50 scfm each that are shared between units 1 and 2. The recombiners are separate and independent. Technical Specification 3.6.4.2. and 4.6.4.2 governs the operability and surveillance requirements. Each recombiner is capable of being manually valved to either containment. The procedures and bases upon which the recombiners would be used will be reviewed in considering shielding requirements and personnel exposure limitations. This effort will be completed in conjunction with the plant shielding review (estimated completion date, January 1, 1980.

TITLE: Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems) for PWR's and BWR's 2.1.6.a.

NRC POSITION

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate Leak Reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction

   Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

RESPONSE

A prerequisite to the establishment of the requested leakage program is the identification of those systems located outside containment which are likely to contain primary coolant immediately following an accident. This will include portions of the accident mitigation systems located outside containment (Safety Injection System, Recirculation Spray System).

We are currently conducting a review of other systems to identify any additional systems which may be used in response to or recovery from an accident. A similar parallel review of systems use under post accident conditions is being conducted by Westinghouse for the Owners Group. It is expected that these reviews will be completed by January 1, 1980.

A leak reduction program for the accident mitigation systems will be developed and implemented by January 1, 1980. If the above mentioned reviews indicate the need for leakage reduction programs in other systems, these additional leakage reduction programs will be developed by March 1, 1980 and implemented as soon as operation permits.

Following is relevant information on the location and functions of the Residual Heat Removal System, Chemical and Volume Control System, Safety Injection System, and Recirculation Spray System. The Residual Heat Removal System (RHR) is located entirely within the containment. The RHR system does not perform any ESF functions. The Chemical and Volume Control System (CVCS) is leak tested as a portion of the Reactor Coolant System (RCS) and is isolated by a safety injection actuation. This leakage testing is governed by Technical Specification

Table 4.1.2.A and is performed at least once each day. Portions of other systems such as the Safety Injection System (SI), and the Recirculation Spray System (RS) are located outside the containment. The RS system is located within the Safeguards Building and the containment where the need for personnel access would be minimal. The outside recirculation spray system is normally aligned for operation and forms a part of the containment boundary. Since North Anna uses a subatmospheric containment, any significant leakage in the outside recirculatin spray system would be obvious during operation due to its impact on establishing or maintaining a vacuum. The Low Head Safety Injection (LHSI) pumps are also located within the Safeguards Building where LHSI lines, when in the recirculation phase, are returned directly to the containment without traversing another building. The LHSI lines to the charging pumps (or High Head Injection Pumps) suction are directed through portions of the Auxiliary Building where the charging pumps are located.

TITLE: Plant Shielding Review (Section 2.1.6.b)

NRC POSITION

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4, each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls.

RESPONSE

A plant shielding review will be performed to evaluate the ability to operate essential systems required after a LOCA with significant core damage. This review will evaluate activity levels in vital areas of the plant which must be accessible to permit operation of essential systems and to ensure that safety grade equipment can perform its intended function in the resulting radiation field. Design changes, increased permanent or temporary shielding and/or post accident procedural controls will be implemented, where required, to assure the proper operation of vital systems.

As with the leakage reduction program, a prerequisite to the completion of the shielding review is the identification of those systems which are likely to contain highly radioactive materials following an accident. As explained in our response to item 2.1.b.a., internal and Owners Group studies are underway to determine which systems, in addition to mitigation systems should be included within the scope of the shielding and leakage reviews.

We are preceding with radiation level calculations for the accident mitigation systems (safety injection and recirculation spray). If the ongoing reviews indicate the need for a shielding review of additional systems, calculations for those system will follow. The identification of any shielding requirements or other corrective actions will be made following the completion of all radiation level calculations.

We expect the shielding review and identification of corrective actions to be completed by March 31, 1980. Implementation of corrective actions will begin by January 1, 1981.

TITLE: Auto Initiation of Auxiliary Feed (Section 2.1.7.a)

NRC POSITION

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term:

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses.

5. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

6. The a-c motor driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

RESPONSE

1. The current design of the Auxiliary Feedwater System provides for automatic initiation.

2. All initiation signals and circuits are designed to prevent a single failure from causing a loss of the Auxiliary Feedwater System.

3. The Auxiliary Feedwater System is initiated automatically by a safety injection signal, loss of offsite power, and on low-low steam generator level of any one steam generator. These actuation signals are testable and these signals are the system actuations on which the FSAR Chapter 14 accident analysis is based. The Auxiliary Feedwater System is also automatically initiated on loss of the main feedwater pumps in anticipation of low steam generator level. This anticipatory actuation is not testable during normal operation.

4. All initiating circuits which automatically start the Auxiliary Feedwater System, are powered from vital buses and are backed-up by the emergency power system.

5. The capability presently exists to manually initiate the Auxiliary Feedwater System from the control room. A single failure in the manual circuits will not result in the loss of system function.

6. The AC motor feed pumps in the Auxiliary Feedwater System are automatically initiated. The motor operated valves and air operated hand control valves required to establish a flow path from the discharge of these pumps to the steam generators are left in the open position and do not receive automatic signals. These valves are under strict administrative control and can be operated from the control room. Therefore, an automatic signal is not required. To further alert the operator if one of these valves is shut, an alarm will be added in the control room which will alarm if any of these normally open valves are not full open. The motor operated valves are powered from the emergency bus. The air operated hand control valves are powered by the vital bus. The valve controls will be powered from a vital bus.

   The AC motor driven pumps are protected by automatic pressure control valves in the pump discharge lines. These valves prevent destructive pump runout in the event of a break in the pump discharge line. These valves are normally closed and automatically open on pump startup to maintain the pump discharge above 900 psig. To alert the operator that the pressure control valves may not be functioning properly after pump startup, an alarm will be added in the control room which will alarm if these valves have not opened.

   All attempts will be made to install the modifications prior to Janaury 1, 1980.

7. The automatic signals are designed in such a manner that their failure will not result in the loss of manual capability to start the Auxiliary Feedwater System.

8. The automatic initiation circuits are presently safety-grade equipment and meet the long-term requirements.

TITLE: Auxiliary Feed Flow Indication (Section 2.1.7.b)

NRC POSITION

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

RESPONSE

1. Auxiliary Feedwater Flow indication to each steam generator which is displayed in the control room is safety grade equipment with the exception of power supply diversity.

2. Auxiliary Feedwater flow indication is powered from the emergency bus via the semi vital bus, which does not meet the diversity requirements of ASTB 10-1 of the standard review plan Section 10.4.9. To meet the diversity requirements of ASTB 10-1 the Auxiliary Feedwater flow indication power supples will be moved to an existing cabinet which meets the diversity requirements. This change will meet the position and will be implemented by January 1, 1980.

3. This change will result in safety-grade indication in the control room of auxiliary feedwater flow to each steam generator and thus satisfies all the requirements, including implementation category B, in NUREG-0578.

TITLE: Improved Post Accident Sampling Capability (Section 2.1.8.a)

NRC POSITION

A design and operational review of the reactor coolant and containment atmosphere sampling system shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremeties, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design change features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperature), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analysis shall be capable of being completed promptly; i.e., the boron sample analyses within an hour and the chloride sample analysis within a shift.

RESPONSE

A design and operational review of the reactor coolant and containment atmosphere sampling will be performed to determine the capability of personnel to promptly obtain samples under accident conditions without incurring a radiation exposure to any individual in excess of limits specified in 10 CFR 20. If this review indicates that samples cannot be promptly and safely obtained, we will provide a description of proposed modifications by January 1, 1980.

A design and operational review of the radiological spectrum analysis facilities will be performed to determine the capability of promptly quantifying core radioisotopes. If this review indicates that the required analysis cannot be performed promptly with the existing facilities, we will provide a description of proposed actions to meet the criteria by January 1, 1980.

These reviews represent preliminary reviews of general sampling capability. We have not addressed certain specific analysis stated in your position because we believe that additional clarification and discussion is needed to identify specific post accident time and sampling requirements. For example, determination of pH should be considered in preference to chlorides. We suggest that these requirements can best be addressed in a topical meeting with the Owners Group.

TITLE: Increased Range of Radiation Monitors (Section 2.1.8.b)

NRC POSITION

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

1. Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

   a. Noble gas effluent monitors with an upper range capacity of 10^5 uCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

   b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from a minimum of 10^{-7} u Ci/cc (Xe-133) to a maxmimum of 10^5 uCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors shall overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by absorption on charcoal or other media, followed by onsite laboratory analysis.

3. In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

RESPONSE

Noble gas effluent and radiation level monitors capable of providing the extended ranges recommended in Section 2.1.8.b do not appear to be commercially available at this time. Studies are being performed, however, to determine appropriate, conservative monitoring ranges based on plant specific parameters. These studies may show that commercially available equipment which approaches but does not reach the recommended range extremes can provide the necessary monitoring capability with an adequate margin of conservatism. Additional studies will be performed to determine how best to utilize existing station equipment for monitoring of radioiodines in gaseous effluents under accident conditions.

The findings of the above studies will be implemented to provide appropriate monitoring capability during the first refueling or extended outage following Jan. 1, 1981 or sooner if possible.

TITLE: Improved In-Plant Iodine Instrumentation (Section 2.1.8.c.)

NRC POSITION

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration throughout the plant under accident conditions.

RESPONSE

Improved capability for the assessment of in-plant airborne radioiodine concentrations under accident conditions will be provided by the following measures.

1. An adequate stock of "silver zeolite" sampling cartridges will be purchased and maintained at the station for emergency use. Existing station equipment will be used to perform gamma spectral analysis on collected samples to accurately assess iodine concentrations.

2. Procedures will be revised to instruct appropriate personnel in the proper precautions to be taken when sampling with charcoal cartridges. The procedure will address acceptable methods for removal of noble gases from charcoal cartridges, prior to performing gamma spectral analysis.

These measures will be implemented by January 1, 1980.

TITLE: Analysis of Design and Off-Normal Transients and Accident (Section 2.1.9)

NRC POSITION

Analyses, procedures, and training addressing the following are required:

1. Small break loss-of-accident accidents;
2. Inadequate core cooling; and
3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).

2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in this appendix).

The analysis of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analysis shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, in the long term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator action is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors shall be put in the form of emergency procedure guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to date, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

RESPONSE

Analyses of small break loss-of-coolant accidents, symptoms of inadequate core cooling and required actions to restore core cooling, and analyses of transient and accident scenarios including operator actions not previously analyzed are being performed on a generic basis by the Westinghouse Owners Group. The small break analyses have been completed and were reported in WCAP-9600, which was submitted to the Bulletins and Orders Task Force by the Owners Group on June 29, 1979. Work required to address the other two areas, inadequate core cooling and other transient and accident scenarios, is being performed in conjunction with the Bulletins and Orders Task Force. Definitions of requirements and schedules for submittal of program results are being established with the Task Force. The results of these programs will determine if the need exists for any additional instrumentation or controls as required by Item 2.1.3.b.

In addition to the above program, the Owners Group is providing pre-test predictive analysis of the LOFT test program in accordance with the schedule established by the Bulletins and Orders Task Force.

The information from these analyses will be included in the plant emergency procedures. The initial Westinghouse review and revision of generic emergency procedures has been completed. The revised procedures were mailed to the NRC on October 16, 1979. Review of the revised procedures by Vepco training and operations personnel is currently in progress. Following acceptance of these procedures by the NRC, station specific procedures will be revised to incorporate the new generic guidelines. Operator training will be updated to include the bases and specifics of the revised procedures.

Any additional procedural changes determined to be necessary as a result of ongoing analyses will be implemented in a timely manner.

Our responses concerning containment pressure, water level, and hydrogen instrumentation and reactor vessel venting are included in Section B1.

TITLE: Shift Supervisors Responsibilities (Section 2.2.1.a)

NRC POSITION

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsiblity of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsiblity and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

   c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities and authority shall be clearly specified.

3. Training programs for shift supervisor shall emphasize and reinforce the responsiblity for safe operation and the management function the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsbility for assuring the safe operatin of the plant shall be delegated to other operations personnel not on duty in the control room.

RESPONSE

1.) A directive will be issued by the Vice President-Production, Operations and Maintenance which emphasizes the primary management responsiblity of the shift supervisor for safe operation of the plant under all conditions and clearly establishes his command duties. This directive will be issued prior to January 1, 1980 and at approximately yearly intervals thereafter.

2a,b,&c.) Existing administrative procedures delineate the responsibilities of station supervisory and operations personnel, including the authority of the shift supervisor. These procedures will be reviewed and revised by January 1, 1980 to include or emphasize the points cited in your position.

In addition, we will increase shift staffing by adding a SRO to allow the shift supervisor to maintain a broader oversight of plant safety and operating conditions. Shift staffing will then include two SRO's for one unit operation and three SRO's for two unit operation.

At least one of these senior reactor operators will be in the control room at all times. The shift supervisor will maintain an overview of plant conditions, make decisions regarding plant operations, and direct the actions of the control room operators.

3. Training programs for shift supervisors and SRO's will be improved by January 1, 1980 to provide greater emphasis on and reinforcement of the responsibility for safe operation and the management function the shift supervisor is to provide. Additional details on this training are included in our response to your position on Shift Technical Advisors.

4. The administrative duties of the shift supervisor will be reviewed by our Director of Nuclear Operations. This review will be completed by January 1, 1980. Additional control room manning, as discussed above, will allow the delegation of routine administrative duties to other control room personnel.

   As explained in response to Item 2.2.1.b., the extra SRO is being added so that an SRO-Shift Technical Advisor will be on each shift. The additional SRO on each shift will begin on January 1, 1980.

TITLE: Shift Technical Advisor (Section 2.2.1.b)

NRC POSITION

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The shift technical advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The shift technical advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the shift technical advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

RESPONSE

As explained in Enclosure 3 of your September 13, 1979 letter, improvements are needed in the following two general areas:

1) Accident assessment capability
2) Review and utilization of operating experience.

Accident Assessment Capability

As explained in response to Item 2.2.1.a we intend to increase shift staffing to include an additional SRO. This additional staffing will begin on January 1, 1980. During 1980 a portion of the station SRO's will begin an advanced training program to qualify them as shift technical advisors. During 1980 least one SRO who is participating in this training program will be on each shift and will be designated as the shift technical advisor (STA). In the event of an accident the shift technical advisor will withdraw from normal operational duties and will be dedicated to assessing plant conditions and advising the shift supervisor.

The shift technical advisors will complete a training program including the following subject:

1. Thermodynamics
2. Fluid Mechanics
3. Heat Transfer
4. Fuel Element Metallurgical Characteristics
5. Natural Circulation
6. Reactor Cooling Alternatives
7. Transients/Accidents* - performed on the Surry Simulator (including, but not limited to)
   a. FSAR accidents
   b. Low RCS pressure
   c. Low RCS flow
   d. Loss of coolant
   e. High RCS temperature
   f. High reactor power
   g. Uncontrolled cooldowns

* Will include multiple failures

It is estimated that this program will involve approximately 160 hours of classroom instruction and an equal time in independent study. Specific course details including content and scheduling will be finalized by January 1, 1980.

During 1980, those SRO's in training for shift technical advisor will complete the above training program so that beginning in January 1981 each shift will be manned with a fully trained shift technical advisor. If during 1980, due to manpower constraints an SRO/STA in training is not available, we reserve the option of substituting a degreed engineer with at least two years of nuclear power plant experience.

In the longer term we intend to upgrade SRO training to include the above training so that all SRO's will be qualified as shift technical advisors.

Review and Utilization of Operating Experience

Beginning not later than January 1, 1981, an individual with the appropriate technical knowledge and operating experience will be assigned full time to the review of internal and industry operating experiences. This person will initiate improvements to plant facilities or operational practices based on the knowledge and experience gained at our stations and throughout the industry. This individual will insure that the proper personnel including the STA's and training personnel receive the necessary experience reports and implement the lessons learned from them.

During 1980, due to manpower constraints, this function will be performed by the Operating Supervisor.

TITLE: Shift and Relief Turnover Procedures (Section 2.2.1.c.)

NRC POSITION

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of availabilty and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included on the checklist);

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic indepenent verification of system alignments).

RESPONSE

Presently, Administrative Procedures outline the requirements for shift turnover. This procedure will be revised by January 1, 1980 to incorporate signed off checklists to verify that systems important to safety are not in a degraded mode. The checklist will verify that primary plant parameters are in a normal band, and system alignments (ie. breaker controls and valve switches) are in accordance with the requirements of the mode of operation which the Unit is in. The checklist will require that the Minimum Equipment Status Record be reviewed with a verification of the time remaining until the Limiting Condition of Operation must be satisfied or the mode of operation changed.

Checklists will be implemented for the auxiliary operating stations which will require that they list maintenance activities in their area on safety related systems which could affect plant operation.

The checklists will require the signature of the oncoming and offgoing operator in each area and the oncoming shift supervisor. These checklists will be implemented by January 1, 1980.

The company quality assurance department conducts periodic audits and inspections to verify compliance with administrative controls including shift turnover procedures.

TITLE: Control Room Access (Section 2.2.2.a.)

NRC POSITION

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

RESPONSE

Existing administrative procedures establish the line of authority from the Station Manager through the Superintendent Operations, Operating Supervisor and the Shift Supervisor. The procedure is explicit in that it states that "the shift supervisor has the responsiblity of directing the actions of the station operators, to ensure safe and prudent operation of the facility". The Superintendent Operations, Operating Supervisor, Shift Supervisors and the Assistant Shift Supervisors are SRO's as mandated by Technical Specification 6.2.2.

The administrative procedures will be changed by January 1, 1980 to establish that the Shift Supervisor or Assistant Shift Supervisor has the authority and the responsibility to limit access to the control room during normal as well as emergency situations. The administrative procedure will establish clear lines of authority and responsiblity during emergency situations. Those persons in charge of the control room and with the authority to direct control room operations, shall be limited to Senior Reactor Operator licensed persons. The administrative procedure will clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

The administrative procedure will clearly limit access to those individuals responsible for direct operation of the station plus other technical advisors as needed by operations. Technical advisors, including non-Vepco personnel will be limited in number during any emergency or abnormal conditions.

These administrative procedure changes will be completed by January 1, 1980.

TITLE: Onsite Technical Support Center (Section 2.2.2.b.)

NRC POSITION

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center.

Records that pertain to the as-built conditions and layout of structures, systems, and components shall be stored and filed at the site and accessible to the technical support center under emergency conditions. Examples of such records include system descriptions, general arrangement drawings, piping and instrument diagrams, piping system isometrics, electrical schematics, wire and cable lists and single line electrical diagrams. It is not the intent that all records described in ANSI N45.2.9-19-1974 be stored and filed at site and accessible to the technical support center under emergency conditions; however, as stated in the standard, storage systems shall provide for accurate retrieval of all pertinent information without undue delay.

RESPONSE

The Onsite Technical Support Center will be established in the Records Building. The Records Building is adjacent to the power station inside the protected security area. The Records Building has a complete controlled set of drawings, technical manuals, and other records which are properly stored and accessible. Communications will be upgraded by establishing NRC phone systems in the building as well as placing adequate commercial phones in the Center. The station PA system is also be available in the Center. We are investigating the installation of a typewriter accessing the Unit 1 process computer in the OTSC. This typewriter could trend critical plant parameters for review by the technical support staff and consultants. If possible this installation will be completed by Janaury 1, 1980. We are currently investigating a more versatile permanent data link for the OTSC.

The Emergency Plan and Emergency Plan Implementing Procedures (EPIP's) will be revised to incorporate the role of the Technical Support Center. The Onsite Technical Support Center will be established as soon as possible, but no later than January 1, 1980. The Emergency Plan and EPIP's will be updated to reflect these changes by January 1, 1980.

TITLE: Onsite Operational Support Center (Section 2.2.2.c)

NRC POSITION

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

RESPONSE

Places of assembly at the onset of an emergency are designated by the Emergency Plan Implementing Procedures (EPIP's). The Emergency Director will announce that all persons are to report to their normal place of assembly for accountablity. This plan of action immediately establishes a situation whereby the shift supervisor or the Emergency Director can locate the needed assistance; i.e. H. P. Techs in the H.P. office, instrument techs in the instrument shop, etc.

The procedures will be changed whereby operators not required for control room operations will gather in the plant assembly room unless performing an operations function outside of the control room or otherwise instructed by the Shift Supervisor. Existing procedures now instruct other emergency teams such as the fire brigade and the first aid teams to assemble in the plant assembly room so as to be readily accessible to the Emergency Director. Other support groups such as Health Physics, Instrument Technicians, Chemistry Technicians, Engineers, and the maintenance personnel will assemble in their respective work areas.

Revised EPIP's will include specific instructions that prohibits their entry into the control room unless requested by the shift supervisor. All of the above areas are served by adequate communications including commercial intraplant telephone and the station PA system. An intraplant telephone will be installed in the plant assembly room to allow better communications from the Control Room to the Operations Support Center. A station PA system is already available.

Procedures will be revised to reflect the above changes by January 1, 1980.

Section A3

Response to Enclosure 7 of September 13, 1979 letter, "Near Term Requirements for Improving Emergency Preparedness".

NRC POSITION

(1) Upgrade licensee emergency plans to satisfy Regulatory Guide 1.101, with special attention the development of uniform action level criteria based on plant parameters.

(2) Assure the implementation of the related recommendations of the Lessons Learned Task Force involving instrumentation to follow the course of an accident and relate the information provided by this instrumentation to the emergency plan action levels. This will include instrumentation for post-accident sampling, high range radioactivity monitors, and improved in-plant radioiodine instrumentation. The implementation of the Lessons Learned Task Force's recommendations on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.

(3) Determine that an emergency operations center for Federal, State and Local personnel has been established with suitable communications to the plant, and that upgrading of the facility in accordance with the Lessons Learned Task Force's recommendation for an in-plant technical support center is underway.

(4) Assure that improved licensee offsite monitoring capabilities (including additional thermoluminescent dosimeters or the equivalent) have been provided for all sites.

(5) Assess the relationship of State/local plans to the licensees' and Federal plans so as to assure the capability to take appropriate emergency actions. Assure that this capability will be extended to a distance of ten miles. This item will be performed in conjunction with the Office of State Programs and the Office of Inspection and Enforcement.

(6) Require test exercises of approved emergency plans (Federal, State, local and licensees), review plans for such exercises, and participate in a limited number of joint exercises. Tests of licensee plans will be required to be conducted as soon as practical for all facilities and before reactor startup for new licensees. Exercises of State plans will be performed in conjunction with the concurrence reviews of the Office of State Programs. As a preliminary planning bases, assume that joint test exercises involving Federal, State, local and licensees will be conducted at the rate of about ten per year, which would result in all sites being exercised once each five years. Revised planning guidance may result from the ongoing rulemaking.

RESPONSE

(1) The Emergency Plan that is currently in effect at North Anna Power Station satisfies the requirements of Regulatory Gide 1.101.

(2) Our responses to the Lessons Learned Task Force recommendations regarding instrumentation to follow the course of an accident are included in Section B2 of this response. Existing emergency procedures use currently installed instrumentation in establishing emergency action levels. By January 1, 1981, following installation of additional instrumentation, emergency plans will be revised to include definitive criteria for emergency action levels using the new and existing instrumentation.

(3) The station visitors center has been designated as the emergency operations center for Federal, State and local personnel. This facility will be upgraded in conjunction with the Onsite Technical Support Center. We are currently evaluating locations for an alternate emergency center. An alternate center will be designated by January 1, 1980.

(4) Offsite monitoring capabilities will be reviewed and improved as required by January 1, 1980.

(5) We have reviewed state/local plans and are satisfied with their capability to take appropriate emergency action.

(6) Emergency exercises for site personnel are conducted yearly. Coordinated emergency exercises involving Federal, State, and local agencies will be conducted every 5 years.